AAA Quarterly Review: Fuel Development at ANL

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July 10, 2002

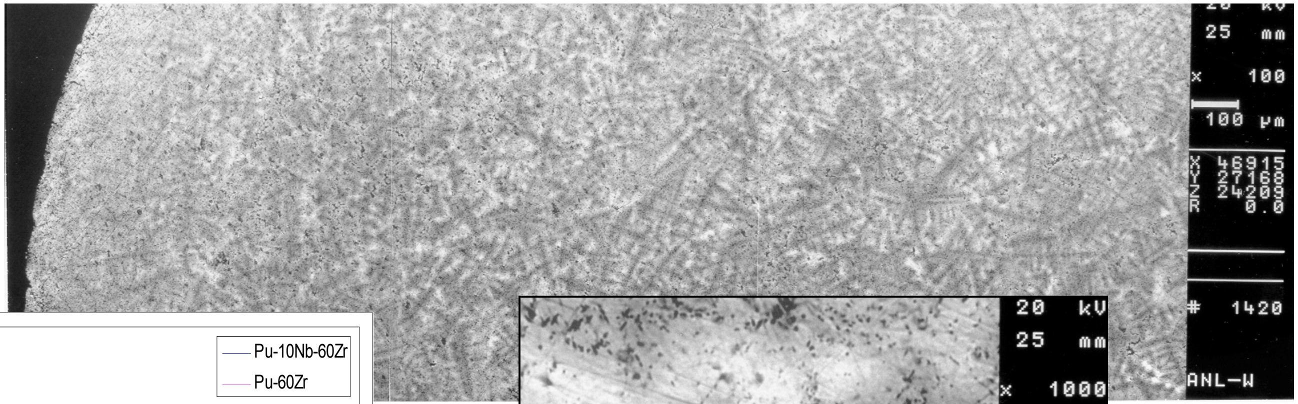
Fuel Fabrication Status

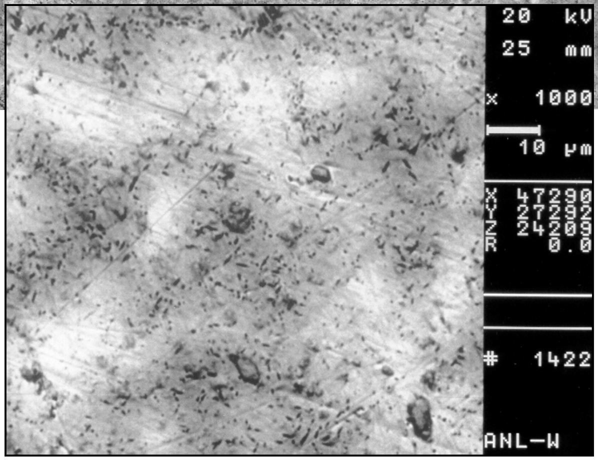
- Fabrication of all metal fuel slugs complete
- Rodlet and capsule welding parameters being refined
  - Target: 100% acceptance
    - Small size of rodlet specimens causes some problems with repeatability
    - QA acceptance ~90%
  - Slight distortion of capsule end caps
  - Corrective actions pursued for both issues

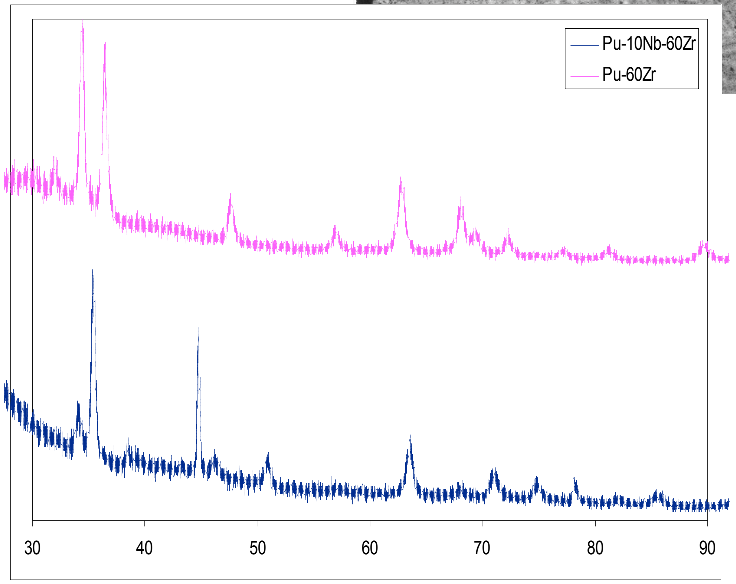
- On target for December 2002 insertion

Metal Fuel Characterization

- Microstructural characterization proceeding
- Example: Pu-10Np-40Zr







Review of Pu-bearing IMF and MOX

- ‘Older’ work
  - Fairly large database
  - Good work on ZrO_2, MgO-based fuels
- More recent Work
  - Paper studies and fabrication
  - No irradiation testing
- Advanced MOX
  - SUPERFACT experiment

Older IMF Work in the U.S. (pre-1970)

- Relevant past work mostly related to 'spike' elements for Pu burning in thermal systems for the Plutonium Utilization Program
  - Thermal Spectrum Fuel Irradiation Tests
    - Al-Pu alloy fuel
    - PuO_2/ZrO_2
    - PuO_2/MgO
- 'Phoenix' whole core demonstrations (reactivity control using ^{240}Pu)
  - Materials Test Reactor (MTR, 1958)
  - Plutonium Recycle Test Reactor (PRTR) + MOX (1963)
- Bettis Atomic Power Laboratory
  - ZrO_2-UO_2 fuels for Shippingport reactor (also CaO-ZrO_2, BeO, Al_2O_3, CeO_2)
- Miscellaneous fuels in thermal spectrum
  - PuN, PuC, PuO_2, PuO_2/graphite, PuO_2 silicate glass
  - Isotope targets, often Al matrix dispersions

Al-Pu Alloys

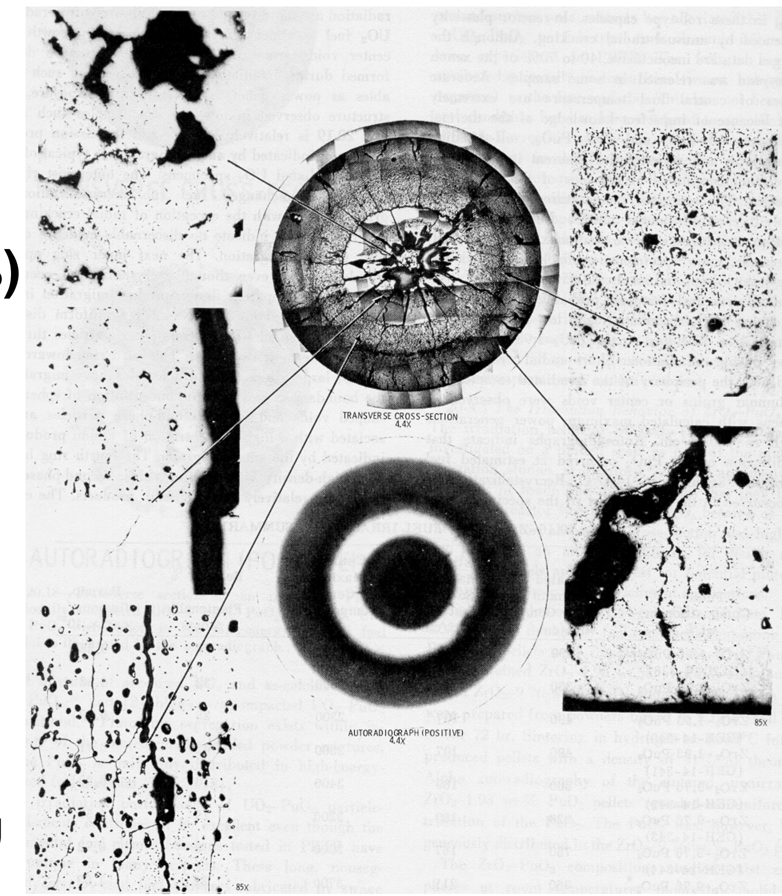
- Al-Pu dispersions similar to early Al-U research reactor fuel
  - Al-Pu eutectic at 15.6 wt% (~2 at%) Pu, 640°C
  - Hypoeutectic fuels ideal for thermal burning
    - 3.35 wt% Pu content gives 95 vol% Al-0.27 wt% Pu matrix
  - Hypereutectic systems also studied
- Fabrication typically by extrusion/coextrusion
- Typically operate at high power density (~100 kW/m)
- Very high Pu burnup possible (90% FIMA)
- Pu-Al segregation must be controlled on fabrication
- Fuel centerline temperature limited to <400°C
- Corrosion resistance improved by Ni, Si, Ti

Al-Pu Alloy Irradiation Testing

- **PRTR (Plutonium Recycle Test Reactor)**
  - **Goal: Suitability of Al-Pu for use in power reactors**
  - **75 elements (1500 rods) 8.26 cm dia. x 2.51 m, 3 failures (1962)**
  - **Zircalloy cladding**
    - » **Fuel/clad gap required due to $\square_T$ mismatch**
  - **Powers to 39 kW/m; fuel center temps. to ~400°C**
    - » **MTR/ETR capsule tests to 520°C, stable □-structure**
  - **Maximum average burnup was 65%/ peak 87%**
- **MTR (Materials Test Reactor)**
  - **Full 'Phoenix' core loading in 1958**
  - **Aluminum clad Al-14 wt% Pu**
  - **Plate-type fuel**
  - **Burnup to 75% FIMA**
- **EBWR (Experimental Boiling Water Reactor)**
  - **10 rods, 3.35 wt% plutonium (8 and 26 wt% ^{240}Pu)**
  - **2 wt% Ni**
- **USAEC HW-69200, IDO-16508, HW-70158, HW-SA-2425**

ZrO_2-PuO_2 Fuels

- **Plutonium Utilization Program**
- **Zircalloy clad 1.44 cm OD specimens in ETR**
- **4 ZrO_2-1.93 wt% PuO_2, 4 ZrO_2-9.76 wt%PuO_2**
  - **Cubic + monoclinic phases**
- **Irradiated in ETR**
  - **Power: 29-95 kW/m**
  - **Temperature: 1400 – 3700°C (±20%)**
  - **Burnup: 8 – 43%**
- **One failure at 95 kW/m**
  - **1/5 of fuel molten**
  - **No loss to coolant**
- **USAEC HW-SA-3128**

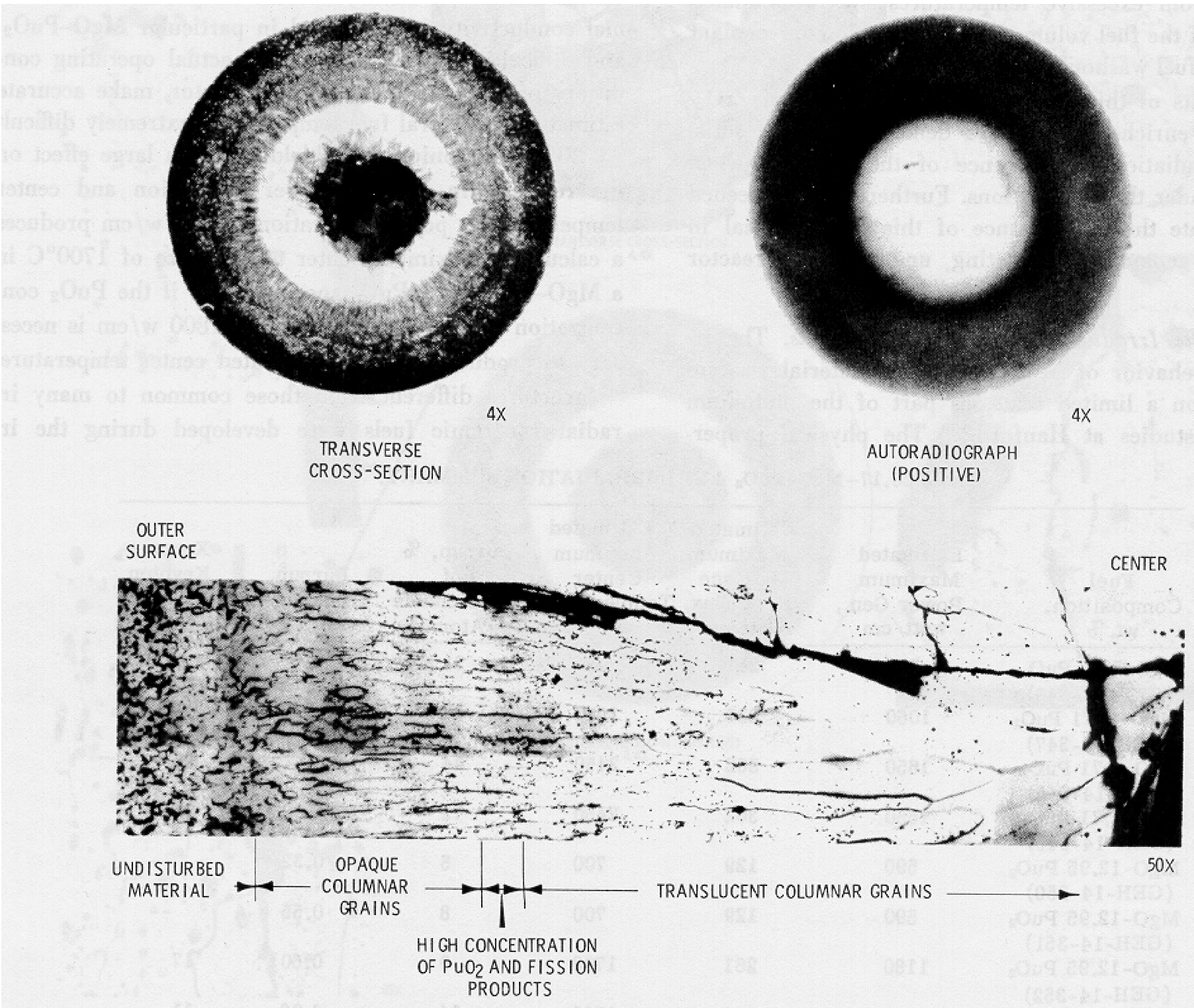


86 kW/m, T_{max} ~3400°C, 10% BU

MgO-PuO_2 Fuel Irradiation Testing

- Zircalloy clad specimens in ETR
  - 4 MgO-2.71 wt%Pu, 4 MgO 12.95 wt% Pu, 1.44 cm OD
  - Sintered 1600°C for 12 hr. in He to 86-92% density
  - Peak power 59-165 kW/m, burnup 5-72%
  - Peak Temperatures 700 – 2450°C (±20%)
    - » Central void and major Pu redistribution at 165 kW/m
- Zircalloy clad specimens in PRTR
  - 19 1.43 cm OD x 251 cm rods, 2.1 wt% PuO_2
  - Swage compacted –6 mesh MgO + -325 mesh PuO_2
  - Failure 3 hours after full power (60 MW)
    - » Irradiation continued 8 days, 23 cm fuel lost
  - Cause: high local temps, F contamination of Pu, water in MgO caused cladding breach.
- USAEC HW-SA-3127, USAEC HW-76300

MgO-PuO_2 Fuel Irradiation Testing



165 kW/m,
2450°C,
72% burnup

More recent work in the U.S. (1970 +)

- **Idaho National Engineering and Environmental Laboratory (INEEL)**
  - Analysis of material properties, neutronics, and fuel performance of Y-$(Zr,Pu)O_2$ pellet fuels (1994)
- **Los Alamos National Laboratory (LANL)**
  - Neutronics calculations, fabrication of small quantity of CaO-$(Pu, Zr)O_2$ pellets, Xe^{2+} and I^{+} ion beam irradiation (late 1990's)

IMFs Proposed at May 02 FDWG Meeting

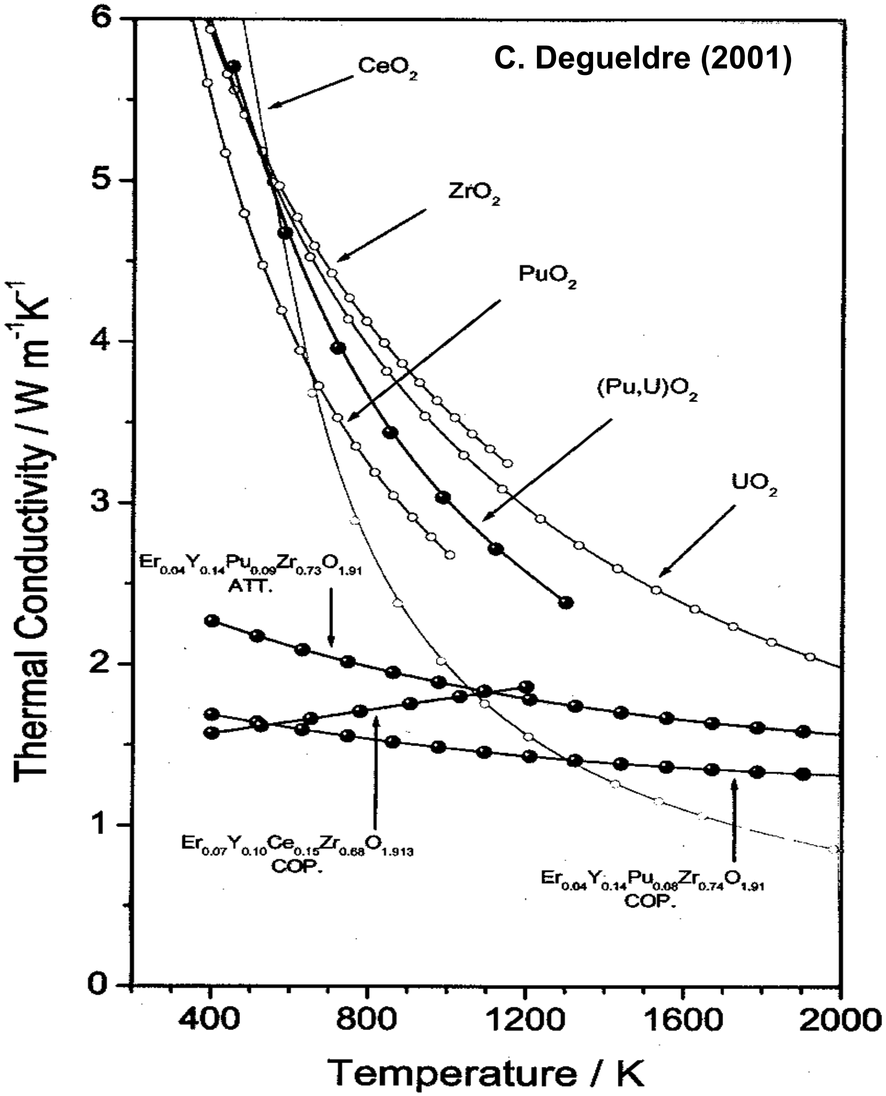
- ZrO_2 solid solution
- MgO-based CerCer
- Zr-matrix Cermet
- SiC-based CerCer
- Ni-Al CerMet

- Also consider Advanced MOX

ZrO_2 solid solution

- **Positive aspects**
  - Good database
    - Out-of-pile data
    - ‘Old’ irradiation data
    - Will soon have new irradiation data with erbia poison (PSI)
    - Current indications of good irradiation performance
  - Easy to incorporate burnable poisons in solution
- **Potential problems**
  - Thermal conductivity ~half of UO_2
    - Reported to be stable with irradiation
    - Power profile shifts to pellet center with Pu depletion
    - Possible solution – annular or filled annular pellets
  - Recycle (?)
    - Slow dissolution, poor solubility in HNO_3
    - Possible solution in pyroprocessing?

ZrO_2 Solid Solution



Thermal conductivity of Zirconia-based fuels is low, but has small dependence on temperature

MgO matrix fuels

- **Positive aspects**
  - Some database
    - » Out-of-pile data
    - » MATINA fast-spectrum data on MgO, $MgO\text{-}UO_2$ (1.2 at% burnup)
    - » 'Old' irradiation data from ETR, PRTR
  - Good thermal conductivity
    - » Indication of 40~60% decrease with neutron irradiation
  - Resistant to melting on high-T accidents (T_m = 2830°C)
  - Recycle - dissolution shouldn't be a problem (?)
- **Potential problems**
  - Solubility in coolant water
    - » PRTR experience
    - » Possible fix – determine mechanism. May be able to 'alloy' to increase corrosion resistance
  - Volatility at high temperatures

Zirconium matrix dispersion fuels

- Positive aspects
  - Some database on similar fuels (stainless steel-based)
  - Fabrication of pins uses fast, simple technique (extrusion)
  - Very low particle volume loading
    - 10-20 vol.% (depending on poison, solid solution)
    - Should be capable of very high burnup
    - Cold fuel - can operate at high power density if required
- Potential Problems
  - Particle coating of (Y-Zr,Pu)O_2 likely to be required
  - Large amount of zirconium in process – impact on recycle
  - Commercial sector acceptance of novel fuel

Dispersion Fuel Performance

(stainless steel matrix)

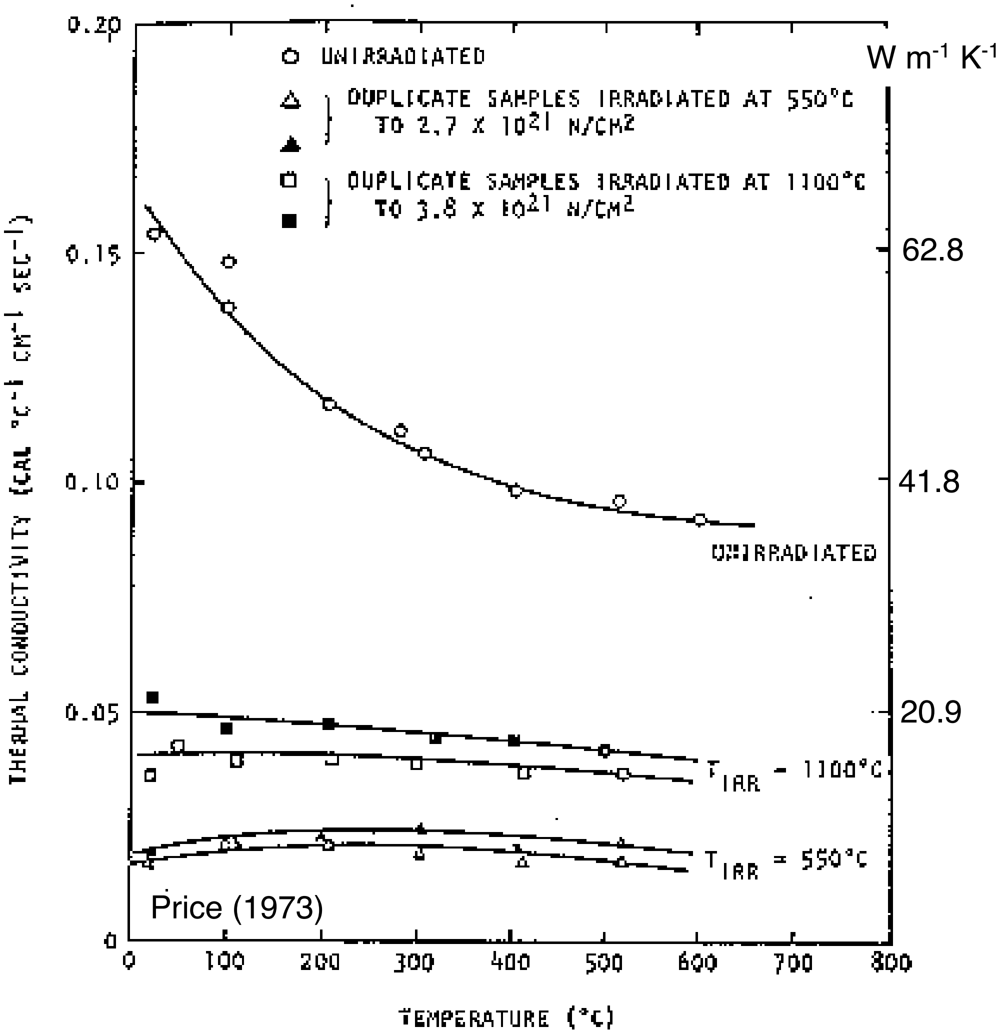
Fuel	Year	Loading, volume %	Surface Temperature	Burnup, % U	Result
UO_2	1960	20	370°C	40-45	$\Delta\rho_{max}$=3%
UO_2	1963	20	538°C	74	full-size plates, some to 84% b.u.
UO_2	1963	27	315-427°C	61	full-size plates, severe cracking
UO_2	1965	30	~620°C	16.2	fast flux
UO_2	1965	50	~620°C	13.5	swelling, but no cladding failure
UN	1960	20	930-1090°C	3.5-5.0	$\Delta\rho_{max}$=1.5%, some blisters

- Heavy metal burnup of 93% enriched fuel
- Plate-type fuel
- Data from UKAEA reports
- Performance depends on microstructure and temperature

SiC matrix fuel

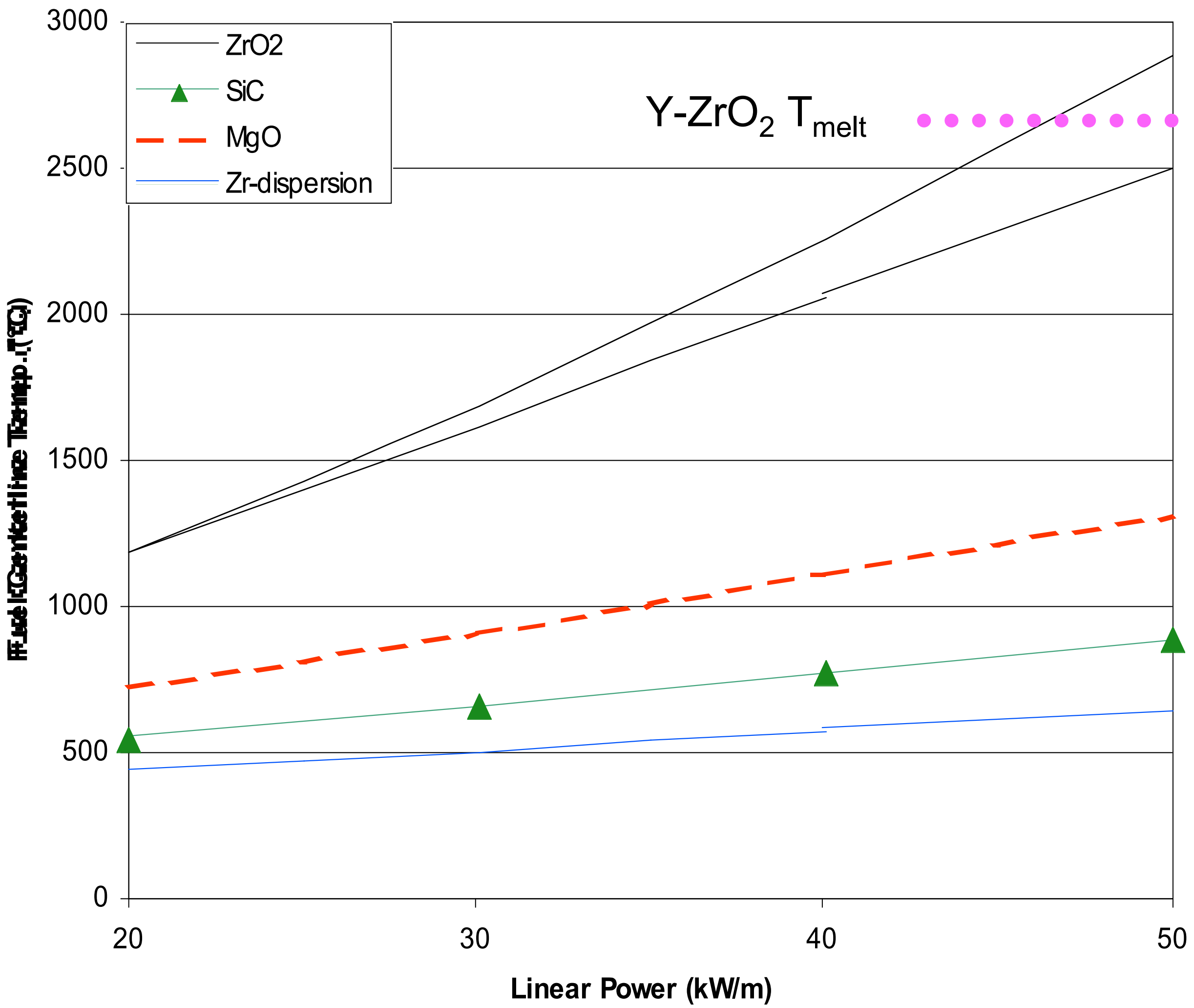
- **Positive aspects**
  - Good thermal conductivity
  - High melting 2700°C (sublimation 2250°C) temperature
  - □-SiC appears to be stable under neutron and H.I. Irradiation
  - Reported good corrosion resistance in acidic and neutral solutions at 290° -320°C
  - Some data relevant to LWR fuel
    - » Fabrication with CeO_2 (Al_2O_3 and Y_2O_3 sintering aids, AECL-1999)
    - » Thermal conductivity measurements of SiC- CeO_2, neutron irradiated pyrolitic SiC
    - » Some recent irradiation data on encapsulated UO_2 pellets in HFR
    - » 72 MeV iodine bombardment produced no swelling (AECL)
- **Potential problems**
  - Recycle may not possible with HNO_3-based process

Thermal conductivity of SiC matrix fuel



- □= 30-100 W m^{-1} K^{-1} at RT
- 5-15 fold reduction at fluence > 2×10^{24} n/m^2
- □□ = f(T)

Examples of Estimated Fuel Centerline Temperatures



Standard 17x17 PWR rod geometry:

- Clad OD = 0.914 cm
- Clad ID = 0.886 cm
- Fuel OD = 0.784 cm
- $T_{coolant}$ = 305°C

16 vol.% SiC, MgO, Zr matrix dispersions. Best guess at $\square_T$.

PWR conditions:

- Avg. power 18-20 Kw/m
- Peak power 43-50 kW/m

Advanced Mixed Oxide Fuels

Superfact experiment – fast reactor fuel in Phénix

Name of the sample	Composition	Density		Origin	O/M	
		g/cm3	% d^{th}		Initial	final
Am 12.4	$(Am_{0.5}U_{0.5})O_{2-x}$	9.70	93%	GSP	1.33	1.81
Am 13	$(Am_{0.5}U_{0.5})O_{2-x}$	10.50	95%	copr	1.33	1.92
AmNp 1	$(Am_{0.25}Np_{0.25}U_{0.5})O_{2-x}$	10.55	95.5%	copr	---	---
Np 2.3.4	$(Np_{0.5}U_{0.5})O_2$	10.50	95%	GSP	2.00	2.00
Np 2.3.5	$(Np_{0.5}U_{0.5})O_2$	10.50	95%	x-GSP	2.00	2.00
Np 3	$(Np_{0.5}U_{0.5})O_2$	10.50	95%	copr	2.00	2.00

Babelot, JRC-ITU-TN-99/03 (1999)

Preparation: 1984-1986

Irradiation: 1986-1988, 324 EFPDs

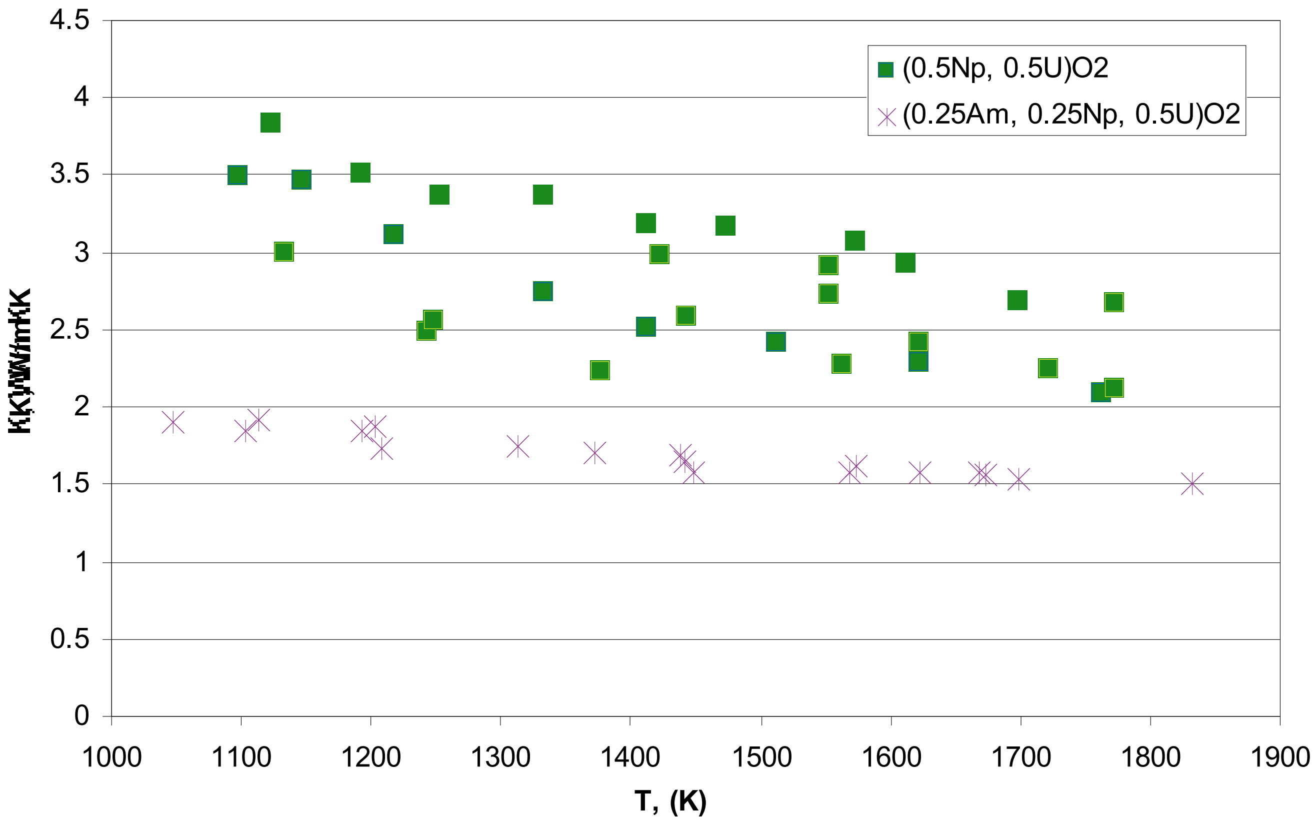
PIE: 1989 -1992

Superfact Experiment

Fuel Composition	MOX -2%Np	20%Am – 20% Np	45% Np
Peak Linear Power BOC (kW/m) EOC (kW/m)	 385 350	 174 286	 197 301
Peak Burnup (%IHM) Np transmutation Am tansmutation	6.6 30.2%	4.1 34.4% 29% (avg.)	4.6 26.3

Note: Purex reprocessing demonstrated extraction of U, Pu and 95% of Np for the composition (U, $Am_{0.2}$, $Np_{0.2}$)O_2

Advanced Mixed Oxide Fuel



- Experimental work on thermal conductivity, oxygen potential, fabrication.
- Basis for extrapolation to LWR fuels.

Babelot, JRC-ITU-TN-99/03 (1999)

Planning for Tier-1 Fuel Development

Short-term: Provide sufficient technical information by the end of FY06 to DOE and/or Congress on the feasibility of fuels for LWR transmutation to support a decision on program continuation.

Long-term: Development and deployment of a fuel cycle designed for rapid destruction of Pu and potentially Np and Am in commercial light water reactors.

Assumptions

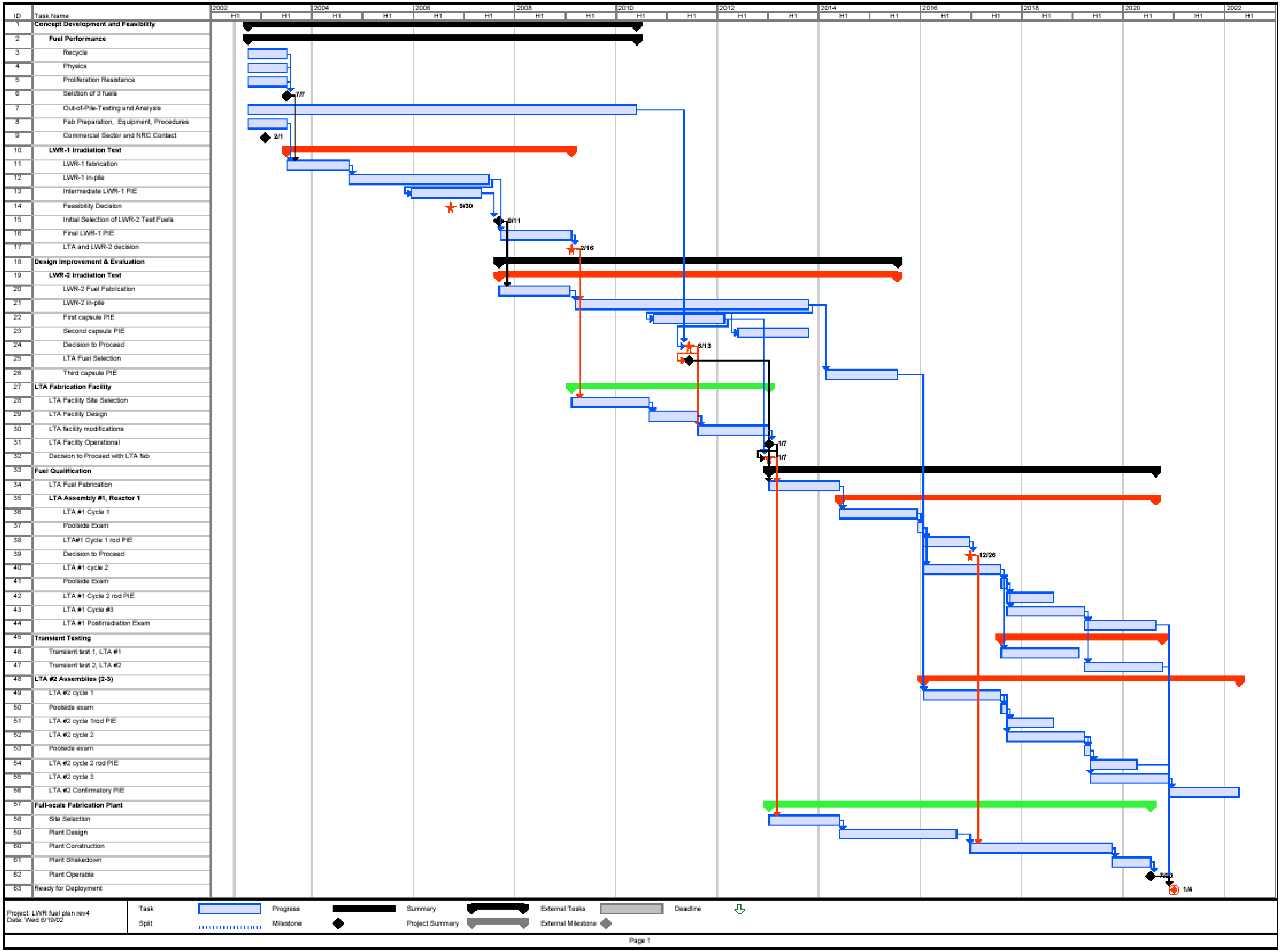
- Deployment should begin within 12-17 year time period (FY15-20) if possible
  - Low risk technology
- Nonproliferation is a key consideration
  - Must mesh with fabrication, inspection, and handling
- Fuel must be compatible with a demonstrated recycle process
- Deployment is in commercial reactors
  - Minimum of new requirements on operations
    - » No additional power or handling restrictions
    - » Reactor safety case not substantially affected
  - Demonstrated accident performance at least as good as UO_2
  - Fuel performance as least as good as UO_2
    - » Should be an economic incentive for operators

‘Five-Year’ Fuel Development Plan Goals

- Provide data for decision in approximately five-years in these areas:
    - » Fuel performance
    - » Fuel recycle (Fabrication)
    - » Core physics
    - » Core safety
    - » Ability to license for use in commercial LWRs
    - » Commercial operator acceptance
- Sort out issue of proliferation resistance and implications on commercial deployment ASAP
- Involve commercial operators/NRC in fuel development process

Long-term IMF Development

- Goal to deploy ASAP drives early schedule
- Deployment possible ~ CY2020
  - Early start on irradiation testing – no substitute
  - Requires steady program
  - Depends heavily on cooperation of the fuel
- Requires some risk
  - Decision points are not optimum - often making technical decisions without complete data
  - Probably requires transient testing



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6 Selection of 3 fuels
7 Out-of-Pile-Testing and Analysis
8 Fab Preparation, Equipment, Procedures
9 Commercial Sector and NRC Contact
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Proposed LWR Fuel 'Five-Year' Plan

- **Evaluate fuel candidates (FY03)**
- **Establish commercial/NRC contacts (FY03)**
- **Fabrication of first fuels for LWR-1 early insertion (FY03)**
- **Out-of-pile characterization (FY03)**
- **Advanced fuel fabrication development (FY03 – FY08)**
  - **Authorization and equipment upgrades (FY03)**
  - **Fabrication experiments on advanced IMFs for LWR-1 (FY04)**
- **Irradiation tests (ATR- FY04)**
  - **LWR-1 scoping test**
    - **Instrumented test facility**
      - **Relatively short duration test**
      - **Power slightly > than prototypic**
    - **2-3 IMF + MOX (AMOX)**
    - **Insertion beginning FY04**
    - **Testing possibly continuing with advanced IMFs**
    - **First PIE mid FY06**
  - **LWR-2 prototype LWR testing to follow in ATR (FY07)**

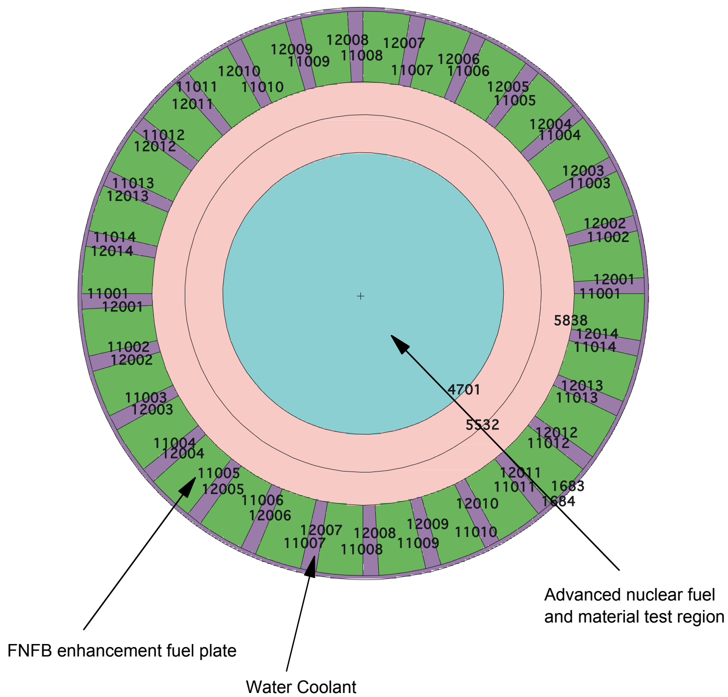
Proposed LWR Fuel Plan: Year One

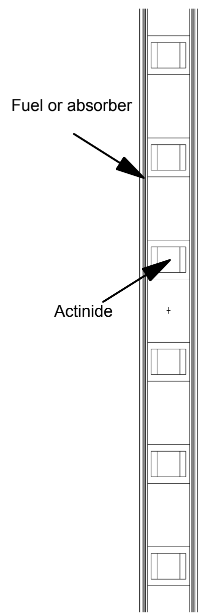
- Spend first 9 months performing initial screening studies/brainstorming (October ’03 – June ’03).
  - Preliminary fuel design concepts
  - Fabrication
  - Fuel performance
  - Recycling flowsheets/ranking
  - Proliferation resistance (ability to incorporate features)
  - Physics analysis
  - Core accident performance
  - Operator acceptance/NRC licensing
- Develop fuel selection criteria
- Objectively rank fuels against criteria
- Select 2-3 IMF for irradiation testing (+ MOX reference and AMOX (?))
- Begin fabrication and irradiation test planning for FY04 insertion

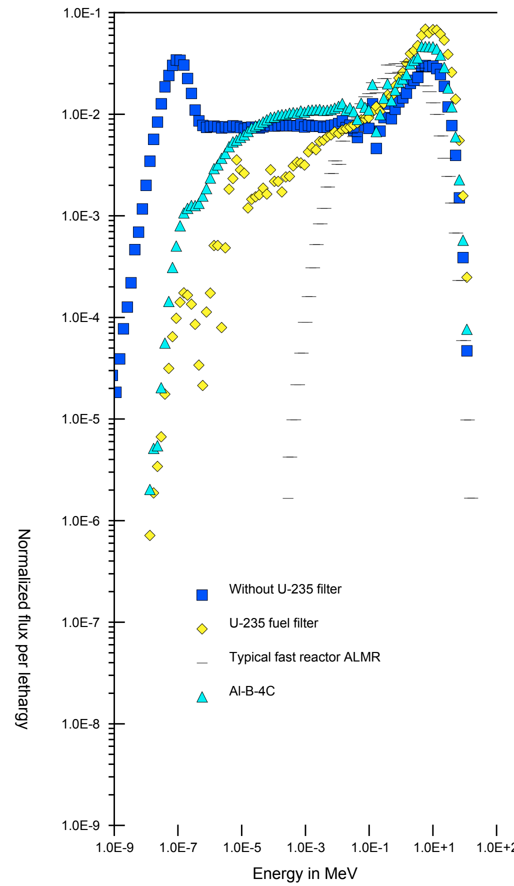
Impact of LWR fuel on Tier 2 Fuel Development

- Tier 2 fuels require a longer lead time for deployment due to:
  - Lack of properties data
  - New fabrication technology required
  - Lack of fuel performance data
  - Difficulties in fast-spectrum testing
  - Undefined deployment scenario and operating conditions
- Tier 2 development should continue to achieve deployment ~ 2030
- Some synergy with LWR fuel development may result in cost savings
  - Scientific and technical personnel
  - Pu fabrication equipment and laboratory facilities
  - Thermal spectrum irradiation testing
  - PIE equipment
  - Transient testing
- Program should continue with some modifications
  - Domestic fast-spectrum test space
  - Potentially fewer fuel choices

ATR Fast Flux Booster

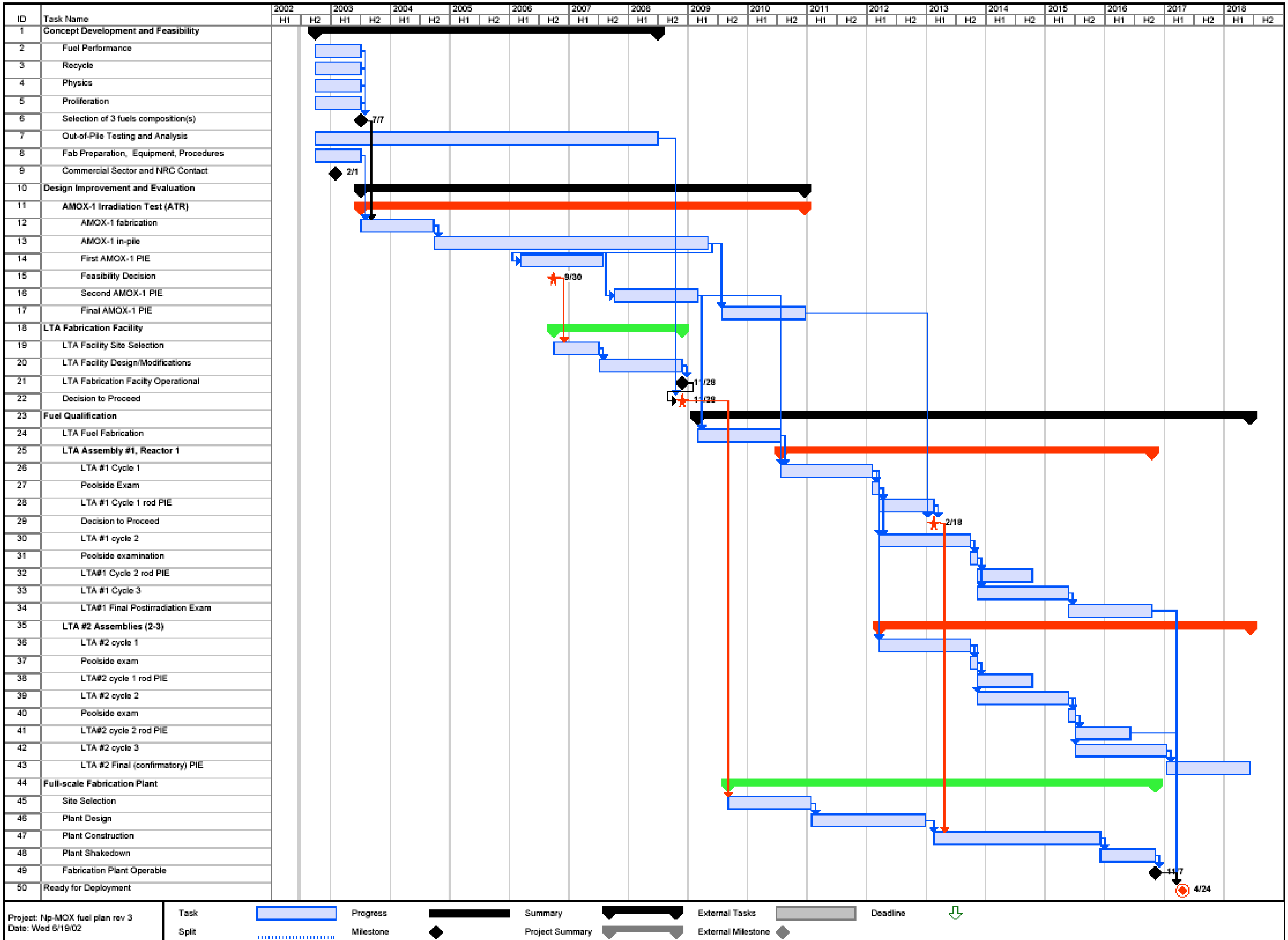






Advanced MOX

- Probably (U, Pu, Np)O_2 (low Am content)
- Shorter lead time for deployment due to:
  - Better properties database (Superfact)
  - Similarity to MOX
    - Some fuel performance data (Superfact)
  - Need for transient testing ?
- Deployment may be possible 2016 ~ 2017
  - One prototype developmental irradiation test prior to LTAs
  - LTA irradiations drive schedule beyond 2008
- Same questions about proliferation resistance
- Should be included in LWR-1 irradiation test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Conclusions

- Does not appear to be a 'perfect' choice of IMF
- Need to provide preliminary data in these areas:
  - » Fuel performance
  - » Fuel recycle (Fabrication)
  - » Core physics
  - » Core safety
  - » Ability to license for use in commercial LWRs
  - » Commercial operator acceptance
- Choice of fuel heavily dependent on current state of technology
  - Could be ready for implementation in 12-17 years
  - Some risk involved in developing new fuels
    - » IMF offers more rapid in-reactor destruction rate than AMOX
    - » Advanced MOX easier, faster to implement